U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.	50-247
License No.	DPR-26
Report No.	96-05
Licensee:	Consolidated Edison Company of New York, Inc.
Facility:	Indian Point 2 Nuclear Power Plant
Location:	Buchanan, New York
Dates:	August 4, 1996, through September 14, 1996
Inspectors:	R. Temps, Senior Resident Inspector B. Westreich, Resident Inspector L. Peluso, Radiation Specialist L. Eckert, Radiation Specialist J. Harold, Project Manager, NRR
Approved by:	Curtis J. Cowgill, Chief Reactor Projects Branch 2 Division of Reactor Projects

EXECUTIVE SUMMARY

Indian Point 2 Nuclear Power Plant
NRC Inspection Report No. 50-247/96-05

This integrated inspection included aspects of licensee operations, engineering, maintenance, and plant support. The report covers a 6-week period of resident inspection; in addition, it includes the results of announced inspections by regional radiation specialists and the results of an evaluation of the new simulation facility (Attachment A).

Operations

Two automatic scrams occurred during the week of August 19, 1996. The inspectors concluded that operators responded well to both reactor trips and the plant also responded well with all safety-related equipment responding as designed to the trips. Balance-of-plant equipment problems were minimal. Troubleshooting activities by maintenance and engineering personnel in determining the causes for each of the two scrams were comprehensive. Overall, both reactor startups and the return to full power operation were well controlled activities.

Maintenance

Various maintenance and surveillance activities related to the two reactor trips were observed. The inspectors determined that all work and surveillance testing observed was performed in accordance with appropriate procedures and was completed satisfactorily. Personnel were knowledgeable of their duties and excellent involvement by the cognizant system engineers was noted on several occasions. The inspectors noted proper control of activities both in the control room and in the field.

Engineering

An open item related to the failure of auxiliary feedwater valve FCV-405A to operate properly was reviewed and closed.

Plant Support

Con Edison, in cooperation with the New York Power Authority, continued to implement an overall effective Radiological Environmental Monitoring Program (REMP) and Meteorological Monitoring Program (MMP), including management controls, quality assurance audits, and quality assurance of analytical measurements. The Offsite Dose Calculation Manual (ODCM) was properly implemented. Audits were effective in assessing program strengths and weaknesses. The REMP and MMP were implemented in accordance with the Technical Specifications, the ODCM, and the Updated Final Safety Analysis Report commitments.

TABLE OF CONTENTS

EXECUTIVE SUMMARY ii

TABLE OF CONTENTS iii

Summary of Plant Status 1

I. Operations 1
- O1 Conduct of Operations 1
  - O1.1 General Comments (71707) 1
  - O1.2 Automatic Reactor Scrams of August 19 and 22, 1996 1
- O2 Operational Status of Facilities and Equipment 3
  - O2.1 Engineered Safety Feature (ESF) Walkdown of 120VAC/125VDC Electrical Buses 3
- O7 Quality Assurance in Operations 4
  - O7.1 Site Nuclear Safety Committee (SNSC) Post-Scram Reviews (40500) 4
  - O7.2 Review of Institute for Nuclear Power Operations (INPO) Report (71707) 5
- O8 Miscellaneous Operations Issues (92700) 5
  - O8.1 (Closed) Licensee Event Reports (LERs) 50-247/96-13-00, Containment Isolation Valve Discrepancies, and 50-247/96-14-00, Loss of Process Monitoring Function During Postulated Fires (Appendix R) 5

II. Maintenance 5
- M1 Conduct of Maintenance 5
  - M1.1 Post-Scram Maintenance, Surveillance and Troubleshooting Activities 5
- M8 Miscellaneous Maintenance Issues 7
  - M8.1 (Closed) Violation 50-247/95-007-01 7

III. Engineering 8
- E2 Engineering Support of Facilities and Equipment 8
  - E2.1 (Closed) Unresolved Item 96-004-02; Auxiliary Feed Pump Flow Control Valve FCV-405A Problems 8

IV. Plant Support 9
- R1 Radiological Protection and Chemistry (RP&C) Controls 9
  - R1.1 Implementation of the Radiological Environmental Monitoring Program 9
  - R1.2 Meteorological Monitoring Program (MMP) 10
- R6 RP&C Organization and Administration 11
  - R6.1 Organization Changes and Responsibilities 11
  - R6.2 Annual Environmental Operating Report 12
- R7 Quality Assurance in RP&C Activities 12

Table of Contents (cont'd)

R7.1 Quality Assurance Audit Reports 12
R7.2 Quality Assurance of Analytical Measurements 13
R8 Miscellaneous RP&C Issues 14
R8.1 (Update) Unresolved Item (URI) 96-03-03 14
R8.2 UFSAR Review 15

V. Management Meetings 15
X1 Exit Meeting Summary 15
X2 Management Meetings 15

ATTACHMENT

Attachment A: Headquarters Audit Report of IP-2 Simulator Facility

Report Details

Summary of Plant Status

The unit started the inspection period in operation at 100 percent power. On August 19, 1996, the reactor automatically scrammed from full power. The unit was restarted on August 21, 1996. On August 22, 1996, the reactor scrammed at low power due to high water level in one of the four steam generators as the result of a mechanical problem with a feedwater regulating valve. The unit was restarted on August 23, 1996, and returned to full power operation for the remainder of the inspection period.

I. Operations

01 Conduct of Operations[1]

01.1 General Comments (71707)

Using Inspection Procedure 71707, the inspectors conducted frequent reviews of ongoing plant operations. In general, the conduct of operations was assessed to be good. Specific events and noteworthy observations are detailed in the sections below.

01.2 Automatic Reactor Scrams of August 19 and 22, 1996

a. Inspection Scope (71707)

On August 19, 1996, the reactor scrammed from 100% power on an indicated loss of flow in the reactor coolant system (RCS) loop associated with the 24 reactor coolant pump (RCP). No actual loss of flow in the loop occurred. During the subsequent return to full power operation, the reactor automatically scrammed on August 22, 1996, on high steam generator level that occurred due to a malfunction of the 23 steam generator (SG) main feedwater regulating valve (MFRV). The inspectors reviewed both events, monitored Con Edison's troubleshooting and repair activities as well as control room and plant activities during the reactor startups, and also attended a post-trip review of the first scram that was conducted by the Site Nuclear Safety Committee (SNSC).

b. Plant Events, Observations and Findings

At 8:41 p.m. on August 19, 1996, the Indian Point Unit 2 reactor automatically scrammed from 100% power. The reactor scram was generated by a single loop loss of flow signal; however, an actual loss of flow condition did not occur, as all four RCPs remained running during and following the reactor scram.

---

[1]Topical headings such as 01, M8, etc., are used in accordance with the NRC standardized reactor inspection report outline. Individual reports are not expected to address all outline topics.

A review of the sequence of events printout revealed that the initiating event for the scram originated from the RCP 24 breaker. At time zero on the printout, an RCP 24 breaker tripped signal was generated; 70 milliseconds later the reactor trip breaker for the A train of the reactor protection system (RPS) tripped, causing the reactor scram. At 676 milliseconds after time zero, RCP 24 breaker indicated not tripped, thus clearing the initiating condition for the scram. All safety-related plant systems responded to the scram as designed; four valves in various balance-of-plant systems did not respond as expected, but had no adverse effect on the scram response activities. Post-scram maintenance on these valves is discussed in Section M1.1.

Con Edison concluded that physical cycling of the RCP breaker was not the cause of the event and focused their troubleshooting activities on the RCP 24 breaker position monitoring circuitry and its input into train A of the RPS. Con Edison was unable to definitively determine the cause of the RPS actuation associated with the 24 RCP. Further discussion of Con Edison's troubleshooting activities is found in Section M1.1.

During the shutdown period, Con Edison performed control rod drop time testing to collect data in response to NRC Bulletin 96-01 (Section M1.1). Following the required post-scram review by the SNSC (Section O7.1) and completion of repair and testing activities, the unit was restarted. The reactor was taken critical at about 4:00 a.m. on August 21, 1996. The inspectors observed startup activities and noted that they were well controlled.

During the power increase and while preparing to place the #21 main boiler feed pump (MBFP) in service (the #22 MBFP was already in service), the operators noted an anomaly in the suction flow indications for the MBFPs. Further questioning and analysis of the situation led the operators to conclude that the discharge check valve for the 21 MBFP pump was not working properly as it appeared to be allowing reverse flow through it. This situation was discussed with plant management. They decided it was prudent to take the turbine off-line and to secure the secondary plant prior to removing the 21 MBFP from service so as to minimize the potential for an adverse feedwater system transient. The 21 MBFP discharge check valve (BFD-1) was subsequently opened and inspection revealed that the valve hinge pin was missing. The hinge pin was subsequently located in the 26A feedwater heater along with a second hinge pin, later confirmed to be from the discharge check valve for the 22 MBFP. (See Section M1.1 for a discussion of the repair activities associated with these valves)

Following removal of the 21 MBFP from service, power ascension resumed. Operators had completed the transfer from manual to automatic control on the main feedwater regulating valves (MFRVs) for steam generators (SG) 21, 22, and 24, and were in the process of transferring control on the 23 SG MFRV when a malfunction caused this MFRV to go full open. This caused a feedwater flow excursion to the 23 SG and caused water level to rapidly increase to the point that a high water level trip was generated, resulting in an automatic trip of the reactor, from 21% power, at approximately 4:00 a.m. on August 22, 1996. All safety-related systems responded as designed on the reactor trip.

Following the trip, Con Edison started troubleshooting the anomalous behavior of the 23 SG MFRV. Operator error was ruled out as a cause of the event as the controller for the valve at the time of the incident was in manual with valve demand in the full closed position. The feedwater flow chart recorder trace for the 23 SG showed a sharp spike upward in feedwater flow indicating the MFRV had rapidly gone to the full open position. Con Edison concluded that the malfunction was most likely caused by a problem with the electro-pneumatic (I/P) controller to the valve positioner. (See Section M1.1 for further discussion of Con Edison's troubleshooting activities)

Following corrective actions for the 23 MFRV and repairs to the 21 MBFP check valve, the unit was restarted. The reactor was taken critical at 12:05 a.m. on August 23, 1996, and returned to full power operation for the remainder of the inspection period.

c. Conclusions

The inspectors concluded that operators responded well to both reactor trips and that the plant also responded well with all safety-related equipment responding as designed to the trips. Troubleshooting activities by maintenance and engineering personnel in determining the causes for each of the two scrams were comprehensive (see Section M1.1). The inspectors assessed that the operators showed an excellent questioning attitude when they noted a suction flow rate anomaly when placing the 21 MBFP into service and diagnosing that there was a problem with the associated discharge check valve. Plant management showed conservative decision making in taking the unit off-line prior to securing the 21 MBFP so as to limit any subsequent plant transient. Overall, both reactor startups and the return to full power operation were well controlled activities.

O2 Operational Status of Facilities and Equipment

O2.1 Engineered Safety Feature (ESF) Walkdown of 120VAC/125VDC Electrical Buses

a. Inspection Scope (71707)

The inspector performed an ESF system walkdown on the 120VAC/125VDC electrical system using procedure check-off-list (COL) 27.1.6, Rev. 7, "Instrument Buses, DC Distribution and PA Inverter." Utilizing the COL and electrical drawings, the inspector verified that electrical breakers were aligned in accordance with the COL.

b. Findings, Observations and Conclusions

The inspector determined that overall, the supply breakers for various safety related equipment were aligned per the positions stated in the COL. Two apparent discrepancies were noted by the inspector and were resolved satisfactorily with the Senior Watch Supervisor.

The first involved the position of a breaker in 120V AC distribution Panel 1 (EFP6). The breaker for circuit number 17 was found in the OFF position contrary to the ON position specified in the COL. Circuit 17 supplies control power to the residual heat removal (RHR) pumps 21 and 22 surveillance instrumentation cabinet. Upon further investigation it was determined that the instrumentation is not used during power operations as the instrumentation is used to monitor characteristics of the 21 and 22 RHR pumps when they are in operation during reactor coolant system (RCS) draining and filling. Operation of the system, including manipulation of breaker 17, is controlled by a system operating procedure (SOP). COL 27.1.6 was completed during RCS filling and venting, during the last refueling outage, at which time the breaker for circuit 17 was in the ON position. During plant heatup following the refueling outage, the RHR system was removed from service and the control power for the RHR instrumentation was removed by placing breaker 17 in the OFF position as directed by the SOP. Therefore, the as-found OFF position was consistent with the switch position as left by the SOP.

The second apparent discrepancy was found in the 118 VAC instrument bus 24 located in the central control room (CCR). The breakers for circuits 23 and 24 were found in the OFF position contrary to the ON position specified in the COL. Circuits 23 and 24 supply instrument and control power, respectively, to the gross failed fuel detection (GFFD) system. The GFFD system is no longer in use at IP2, therefore the breakers are maintained in the OFF position. The inspector reviewed the last completed copy of the COL and determined that there was an entry noting the difference in breaker position and the basis. The inspector was informed that permanent removal of the system and the associated circuits are the subject of a planned modification (modification #SNX-94-03027-M) at which time reference to the breakers will be removed from the COL and associated drawings.

O7 Quality Assurance in Operations

O7.1 Site Nuclear Safety Committee (SNSC) Post-Scram Reviews (40500)

Operations Administrative Directive (OAD) 23, Post Trip Review and Evaluation Procedure, requires a post-trip review be conducted by the SNSC prior to subsequent reactor startup. The inspector reviewed the OAD-23 report for the August 19, 1996, scram and noted that it contained detailed information on the plant's response to the scram.

The OAD-23 report was reviewed by the SNSC at a post-trip review meeting held on August 20, 1996. Events described in the OAD-23 report were discussed by the SNSC members. The engineering group provided detailed information on the extent of troubleshooting activities performed to identify the cause of the RPS actuation associated with the 24 RCP circuitry. The inspector noted especially thorough questioning of the engineering group by the SNSC as to the probable cause of the reactor trip since troubleshooting did not identify a definitive cause for the RPS scram activation. Following completion of the OAD-23 review, the SNSC gave their approval for unit restart.

Following the SNSC meeting, a conference call was held between Region I managers and technical specialists and Con Edison to discuss the results of Con Edison's investigation into the scram. The same information provided at the SNSC meeting was reviewed with the regional personnel. The inspector's assessed that both meetings were well conducted and that Con Edison's decision to restart the unit was made only after a thorough review and discussion of all of the facts pertaining to the August 19, 1996, scram.

O7.2 Review of Institute for Nuclear Power Operations (INPO) Report (71707)

The inspectors reviewed the INPO document containing the site evaluation report for Indian Point 2 that was conducted from October 16 through 23, 1995. The report was issued in March of 1996, and was reviewed soon thereafter by the inspectors. The inspector's review indicated that INPO's assessment of plant activities was consistent with the NRC's perception of Con Edison's performance at the time of the evaluation.

O8 Miscellaneous Operations Issues (92700)

O8.1 (Closed) Licensee Event Reports (LERs) 50-247/96-13-00, Containment Isolation Valve Discrepancies, and 50-247/96-14-00, Loss of Process Monitoring Function During Postulated Fires (Appendix R)

These LERs document issues identified by Con Edison and the NRC in followup to the identification that containment isolation valve PCV-863 was not operated in accordance with procedures. These issues were discussed in NRC Inspection Report 50-247/96-04, Section O4.1, and were the subject of an enforcement conference held on September 12, 1996. Review of these LERs did not identify any new issues.

II. Maintenance

M1 Conduct of Maintenance

M1.1 Post-Scram Maintenance, Surveillance and Troubleshooting Activities

a. Inspection Scope (62707 and 61726)

The inspectors observed various work activities that were conducted following the reactor scrams of August 19 and 22, 1996. Observations were made in the field and in the control room.

b. Observations and Findings

The inspectors observed or reviewed the following maintenance activities that took place following the reactor scrams:

- Following the August 19, 1996, scram (Section O1.1), Con Edison performed comprehensive troubleshooting of the 24 RCP circuitry and its interface with the RPS. All wiring and components from the 24 RCP breaker cubicle to the interface with Train A of the RPS were inspected and tested, and no anomalous equipment conditions were identified that accounted for the perturbation, which lasted less than 676 milliseconds, that caused the RPS to sense that the breaker for RCP 24 had tripped. It was speculated that the trip may have been generated as a result of a high resistance condition (from oxidation) on a relay contact and that actuation of the relay during the scram could have "wiped" the contact clean, thus accounting for fact that no anomalous conditions were found during the extensive troubleshooting effort.

- The inspector observed post-maintenance testing (PMT) on low pressure steam dump valves 1206 and 1207. Maintenance was performed on the valves as they had failed to stroke properly following the August 19 scram. The inspectors observed stroking of the valves from the control room and in the field and assessed that the PMT was well controlled. The inspectors also noted good involvement by the cognizant system engineer.

- The inspector observed maintenance on feedwater heater extraction check valve 6EX3A that was performed under work order (WO) 85102. The valve was diagnosed in July of 1996, as having its disk separated from the shaft. Con Edison had prepared SNSC approved procedures to reduce reactor power and remove the 26 string feedwater heaters from service for an on-line repair of the valve. This activity was planned for the second week of September; however, following the August 19, 1996, scram, Con Edison was able to effect repairs without having to use the special procedures. The inspector observed the removed valve internals and discussed the apparent failure mechanism with the system engineer. The inspector also observed ultrasonic thickness measurements that were taken on the valve body.

- The inspector observed control rod drop testing that was performed following the August 19,1996, scram. The testing was conducted to comply with commitments related to NRC Bulletin 96-01. The inspector observed conduct of the test from the control room and at the rod control cabinet and noted good communication and coordination of activities between personnel at the two locations. Subsequent analysis of the test data indicated no anomalous conditions.

- In response to the second scram that was caused by the malfunctioning MFRV (Section O1.1), Con Edison initiated extensive troubleshooting to determine the cause of the MFRV problem. Electrical and mechanical faults were ruled out following testing. Con Edison then focused on the pneumatic (air operated) controllers associated with the MFRV.

  The controller vendor was contacted and questioned as to whether there were any credible failure mechanisms that would account for the MFRV's action in going full open. The vendor stated that the entry of foreign material into the controller internals could result in the blocking of internal orifices and subsequent internal pressure increase that would cause the associated valve, in this case the 23 SG MFRV, to rapidly open. Con Edison performed an alcohol flush of the 23 MFRV's I/P controller and a small amount of unidentified foreign material was removed. The I/P controllers for the other three MFRVs were also flushed and no foreign material was found in them. Con Edison concluded that the foreign material removed from the 23 MFRV I/P controller was the likely cause of the valve's observed behavior.

- As discussed in Section O1.1, Con Edison performed repairs to the MBFP discharge check valves. The hinge pin for each of the two check valves was identified to be missing (both were located in and removed from the 26A feedwater heater). The hinge pins are held in place by retaining pins that were also missing. Con Edison determined that interference welds used to keep the retaining pins in place were dimensionally inadequate and allowed the pins to rotate and eventually dislodge. After consultation with the valve vendor, the retaining pin interference welds were modified, during repairs to the valves, to prevent this condition from recurring. Con Edison classified the retaining pin failures as a maintenance preventable functional failure (MPFF) under the guidance of the recently implemented Maintenance Rule (10 CFR 50.65).

c. <u>Conclusions</u>

The inspectors determined that all work and surveillance testing observed was performed in accordance with appropriate procedures and was completed satisfactorily. Personnel were knowledgeable of their duties and excellent involvement by the cognizant system engineers was observed. The inspectors noted proper control of activities both in the control room and in the field.

M8 Miscellaneous Maintenance Issues

M8.1 <u>(Closed) Violation 50-247/95-007-01</u>: In-service Inspection (ISI) Qualifications of NDE Level III Personnel.

The inspector performed a review of Con Edison's commitments documented in their response to NRC Violation 50-247/96-007-01 that dealt with an individual's qualification that did not meet the examination requirements of the American Society for Nondestructive Testing Recommended Practice No. SNT-TC-1A.

Con Edison's immediate corrective action was to administer method specific examinations for the NDE Level III individual to meet the requirements of SNT-TC-1A. Further, a review was conducted to evaluate the qualification and certification packages for all NDE personnel who conduct inspections at Indian Point. The inspector noted that the method specific examinations were successfully completed and the qualification and certification packages for NDE personnel were complete and met the requirements of SNT-TC-1A.

In order to preclude recurrence, Con Edison designated a responsible NDE Level III individual to maintain personnel qualifications for all site NDE inspectors. In addition, the position guide for the individual was revised to include this function. Training was provided to all site NDE personnel on the subject of the NRC violation.

III. Engineering

E2 Engineering Support of Facilities and Equipment

E2.1 (Closed) Unresolved Item 96-004-02; Auxiliary Feed Pump Flow Control Valve FCV-405A Problems

In NRC Inspection Report 50-247/96-04 the inspectors reviewed an event that occurred while performing In-service Testing (IST). One of the four flow control valves (FCV-405A) for the steam driven auxiliary feedwater pump failed to open on demand. As a result of previous IST test failures on this valve, Con Edison decided to perform air operated valve diagnostic testing. During that testing, the valve again failed to open on demand. The valve was opened up and inspected. Inspection revealed galling damage and trim package gasket retaining lip damage to the valve.

In order to determine the condition of the other FCV-405 valves, they were opened and inspected. Although no other galling problems were identified, dye penetrant testing indicated trim package cracking on FCV-405B similar to that identified in FCV-405A. The licensee, after consulting with the manufacturer, removed the trim package outer gasket retaining lips on all four FCV-405 valves. The galling on FCV-405A was repaired and all four FCV-405 valves were then tested in a full flow test. A safety evaluation determined that a similar condition, if present in the four FCV-406 valves associated with the 21 and 23 AFW pumps, would not affect valve operability.

This item was left unresolved pending completion of the corrective actions for these valves. To evaluate valve performance in the future, Con Edison plans to install air operated valve (AOV) diagnostic testing equipment. This testing will be performed with the vendor present to evaluate results. Until this modification is installed, weekly testing of the valves is being performed.

Initial metallurgical analysis of the failed components showed that the correct material was used in the valve internals. A failure mode of the retaining lip has not been identified yet; however, the failure of the retaining lip did not affect valve operability. Based on the corrective actions taken to date and the planned AOV diagnostic testing, this item is closed.

IV. Plant Support

R1 Radiological Protection and Chemistry (RP&C) Controls

R1.1 Implementation of the Radiological Environmental Monitoring Program

a. Inspection Scope (84750)

The inspectors observed and assessed Con Edison's capability to implement the Radiological Environmental Monitoring Program (REMP). The inspectors reviewed the REMP procedure manual, visited selected sampling locations to confirm that samples were being obtained from the locations specified in the Offsite Dose Calculation Manual (ODCM), observed licensee personnel perform sampling duties (exchange air filters and charcoal canisters from air samplers), examined the air samplers to determine operability and calibration status, and reviewed the results of the Land Use Census and the most recent revision of the ODCM. The above areas were inspected against Sections 4.11 of the Technical Specifications (TS), the ODCM, and the Updated Final Safety Analysis Report (UFSAR).

b. Observations and Findings

Con Edison continued to maintain the responsibility of the REMP according to the Memorandum of Understanding (MOU), No. 30, Rev. 2, dated 9/25/93. This MOU delineates the responsibilities of Con Edison and New York Power Authority, Indian Point Unit 3, (NYPA) for maintaining the REMP.

The REMP procedure manual included procedures for air, water sampling methods, and gas meter calibration calculations for the air samplers. Con Edison had good procedures that provided the required direction for implementing an effective REMP.

The REMP Procedure NEM-5.103, "Collection, Preparation and Analysis of Air Samples", contains a statement regarding the proper flow rate of the air samplers as 2 cfm. However, the inspectors noted that the air samplers were operating with a flow rate of 3 cfm. Although the air samplers require a minimum volume to operate, the optimal flow should be determined to avoid breakthrough of the charcoal cartridge and ensure charcoal efficiency. Licensee personnel were unable to provide a basis for the flow rate in procedure NEM-5.103 and stated to the inspectors that they would review this matter to determine the optimal air sampler flow rate. The matter is considered an Inspector Follow-up Item **(IFI) 50-247/96-005-01**.

During a previous inspection, (Section 4.3 of NRC Inspection Report No. 50-247/94-19, dated November 23, 1994), the inspector documented exceptions to certain sampling procedures. During this inspection, the inspectors reviewed these procedures and determined that Con Edison's actions were appropriate.

Sample collection commitments were met and samples were collected from the locations specified in the ODCM. The sampling stations included air samplers for airborne iodines and particulates, a composite water sampling station, vegetation and sediment locations, and several thermoluminescent dosimeter (TLD) stations for measurement of direct ambient radiation. The inspectors witnessed the weekly exchange of charcoal cartridges and air particulate filters at selected sampling stations. All observed air sampling equipment was operational and calibrated at the time of the inspection. The TLDs were placed at the designated locations as specified in the ODCM. Vegetation and sediment samples were obtained from the locations specified in the ODCM. The Land Use Census was performed within a year of the previous census. No significant changes to the ODCM that may have reduced the intent of REMP were identified.

c. Conclusion

Based on the above review, direct observations, discussions with personnel, and examination of procedures, the inspectors determined that Con Edison continued to effectively implement the REMP in accordance with the TS, ODCM, and UFSAR commitments.

R1.2 Meteorological Monitoring Program (MMP)

a. Inspection Scope (84750)

The inspectors observed and evaluated Con Edison's MMP to determine whether the instruments and equipment were operable, calibrated, and maintained. The inspectors reviewed the meteorological equipment calibration procedures and results, the backup diesel generator test results, the meteorological system instrumentation upgrade, and maintenance activities. The MMP was inspected against Sections 3.15 and 4.19 of the TS, Appendix 2A of the UFSAR, and Regulatory Guide 1.23.

b. Observations and Findings

NYPA continued to maintain the responsibility of the MMP according to the Memorandum of Understanding (MOU), No. 13, Rev. 1, dated 9/23/93. This MOU delineates the responsibility of NYPA for maintaining the meteorological instrumentation.

Since the previous inspection, NYPA upgraded the meteorological instrumentation. The upgrade included the removal of the translator cards, the replacement of the analog instrumentation with digital equipment, and rewiring sensors directly to the system's two data loggers. The inspectors reviewed NYPA's 50.59 Analysis pertaining to the design change and noted that it considered the impact on the Indian Point 2 UFSAR as well. No discrepancies in the 50.59 review were identified.

The equipment calibration procedures contained sufficient guidance to implement an effective program. The wind speed, wind direction and temperature sensors were calibrated by the vendor using NIST traceable calibration equipment. The semiannual calibration results were within the established acceptance criteria and met the recommendations of Regulatory Guide 1.23.

The primary tower is equipped with wind direction, wind speed, and temperature sensors at the 10, 60 and 122-meter elevations. The inspectors compared the data output from the recorders, data loggers, and computer in the equipment house to the data output in the control rooms of both Indian Point units. The meteorological data were available and the results of the comparison were in agreement.

The primary tower is also equipped with a diesel generator in the event of a loss of power to the primary tower. The inspectors reviewed the monthly and annual diesel generator tests and noted that the results were within NYPA's established acceptance criteria.

The monthly Meteorological Tower reports were reviewed and the inspectors noted that the reports provided good trending and review of maintenance activities. The inspectors also noted that the Meteorological Tower's reliability was high.

c. Conclusion

Based on the above review, direct observations, discussions with personnel, and examination of procedures and records for calibration of equipment, the inspectors determined that (1) the instrumentation upgrade was an excellent initiative to upgrade the Meteorological Monitoring System to increase system reliability and (2) Con Edison continued to effectively implement their limited role in the MMP in accordance with the MOU, UFSAR commitments and Regulatory Guide 1.23 recommendations.

R6 RP&C Organization and Administration

R6.1 Organization Changes and Responsibilities

a. Inspection Scope (84570)

The inspectors reviewed any organization changes and the responsibilities relative to oversight of the REMP and MMP since the previous inspection conducted in November 1994 to verify the implementation of the TS requirements.

b. Observations and Findings

The inspectors noted that the reporting chain was similar to that of the previous inspection.

c. Conclusion

Based on the above review, the inspectors determined that the responsible personnel cognizant in these programs essentially remained the same.

R6.2 Annual Environmental Operating Report

a. Inspection Scope (84570)

The inspectors reviewed the Annual Environmental Operating Report to verify the implementation of the TS requirements Section 6.9.1.5.

b. Observations and Findings

The Annual Reports for 1994 and for 1995 provided a comprehensive summary of the results of the REMP around the Indian Point Unit 2 and Unit 3 sites and met the TS reporting requirements. No omissions, mistakes, obvious anomalous results or trends were noted.

c. Conclusion

Based on the above review, the inspectors determined that Con Edison maintained good management control to implement the TS requirements.

R7 Quality Assurance in RP&C Activities

R7.1 Quality Assurance Audit Reports

a. Inspection Scope (84750)

The inspectors reviewed the Quality Assurance (QA) audit reports against criteria contained in TS requirements, Section 6.5.2.8.

b. Observations and Findings

The following audits were reviewed:

- A95-09I, Environmental Programs
- 95-12-03-A, Radiological Effluent Monitoring

The scope and technical depth of the audits were good and sufficiently assessed the programs for strengths and weaknesses in the REMP. Actions taken as a result of previous audit findings were followed up by the auditors.

The responsible departments responded to these findings and recommendations in a timely manner.

c. Conclusion

Based on the above review, the inspectors determined that Con Edison conducted an audit of sufficient technical depth and adequately assessed the quality of the REMP.

R7.2 Quality Assurance of Analytical Measurements

a. Inspection Scope (84750)

The inspectors reviewed Con Edison's Quality Assurance (QA) Program for analytical measurements of radiological environmental samples including the Interlaboratory Comparison Program required by the TS and ODCM.

b. Observations and Findings

The QA/QC programs of analytical measurements for radiological environmental samples were reviewed to determine whether Con Edison had adequate control with respect to sampling, analyzing, and evaluating data for the implementation of the REMP.

These programs are conducted by Con Edison's and NYPA's contrator, the J.A. Fitzpatrick Environmental Laboratory (JAFEL), located in Fulton, N.Y. The laboratory maintained internal QA programs including environmental split samples, spike samples, and blind samples and supplied reports of QC results to Con Edison for review. When discrepancies were found, reasons for the discrepancies were investigated and resolved. The inspectors reviewed the JAFEL Quality Assurance Reports for 1994 and 1995 which contained the results of the QA programs and noted that the results of the splits and spike samples were within the established acceptance criteria, with few exceptions.

The laboratory participated in the Inter-laboratory Comparison Program (EPA Cross-check Program). The inspectors reviewed the cross-check results for 1995 and noted that results were within the EPA's acceptance criteria. In 1996, the laboratory started to use a vendor laboratory, Analytics, Inc., to continue the Inter-laboratory Comparison Program since the EPA no longer provided this service after December 1995. The inspector reviewed the cross-check results for the first quarter 1996 and noted that the results were within the established acceptance criteria. The inspectors also determined that the program is equivalent to the EPA Cross-check Program. JAFEL plans to use Environmental Management Laboratory (EML) to supplement the Analytics Program. This program is expected to be implemented in September 1996.

Since JAFEL also obtained calibration standards from Analytics, the inspectors questioned if the samples provided for the intercomparison program are independent from the calibration standards. Review of Analytics program revealed that independence was assured since analytics established two separate and independent programs, one for the calibration standards and the other for the intercomparison program.

c. Conclusion

Based on the above reviews and discussions, the inspectors determined that Con Edison continued to implement a good quality assurance program in accordance with regulatory requirements.

R8 Miscellaneous RP&C Issues

R8.1 (Update) Unresolved Item (URI) 96-03-03

In NRC Inspection Report 50-247/96-03 the inspectors determined that UFSAR Section 9.9.2 system description of the logic conditions for automatic actuation of the booster fans and the charcoal filter unit had not been updated to include fire protection. The inspector found that there had been a modification that included installation of smoke detectors to the Control Room Ventilation System (CRVS) automatic actuation logic. The UFSAR described automatic actuation which occurred in the event of a safety injection signal, high radiation, or toxic gas condition, but did not been include the addition of the smoke detectors to the automatic actuation logic. This item was considered unresolved pending further review and assessment of this area.

During this inspection period, the inspectors continued to review the modification the CRVS actuation circuit. The inspectors discussed the issue with engineering personnel and reviewed documentation. Modification ESG-82-07961, which was performed in 1982, added the smoke detectors, toxic gas monitors and radiation detectors to the recirculation mode actuation function of the system. However, the safety evaluation performed for the modification only addressed the addition of toxic gas monitoring and radiation monitoring system. No safety evaluation was performed to address the addition of the smoke detectors as required by 10 CFR 50.59 when making changes to the facility as described in the FSAR.

Following identification of the lack of safety evaluation and UFSAR description to address the smoke detectors, Con Edison issued safety evaluation NS-2-81-172 REV 2 on August 26, 1996. The evaluation determined that no new failure modes were introduced by the addition of the modification and no other issues were identified. Con Edison intends to update the FSAR to reflect the addition of the smoke detector circuitry. This item will remain unresolved pending revision of the UFSAR.

R8.2 UFSAR Review

A recent discovery of a licensee operating their facility in a manner contrary to the Updated Final Safety Analysis Report (UFSAR) description highlighted the need for a special focused review that compares plant practices, procedures and/or parameters to the UFSAR description. While performing the inspection discussed in this report, the inspectors reviewed the applicable portions of the UFSAR that related to the areas inspected. The inspectors verified that the UFSAR wording was consistent with the observed plant practices and procedures and/or parameters.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management at an exit meeting held on October 3, 1996. Con Edison acknowledged the findings presented. The inspectors asked Con Edison whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

X2 Management Meetings

A pre-decisional enforcement conference was held at the Region I Headquarters on September 12, 1996, to discuss potential violations identified in NRC Inspection Report 50-247/96-04. A copy of Con Edison's slides presented at the enforcement conference is attached as Enclosure 2 to this report. Results of the enforcement conference relative to enforcement action will be issued under separate cover letter from this inspection report.

INSPECTION PROCEDURES USED

IP 40500:	Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems
IP 62707:	Maintenance Observation
IP 61726:	Surveillance Observation
IP 71707:	Plant Operations
IP 92700:	Onsite Followup of Written Reports of Nonroutine Events at Power Reactor Facilities
IP 37551:	Onsite Engineering
IP 84750:	Radioactive Waste Treatment, and Effluent and Environmental Monitoring

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

- URI 96-005-01	Determination of Optimal Air Sampler Flow Rate

Closed

- VIO 95-007-01	Inservice Inspection (ISI) Qualifications of NDE Level III Personnel.
- URI 96-004-02	Auxiliary Feed Pump Flow Control Valve FCV-405A Problems
- LER 96-13-00	Containment Isolation Valve Discrepancies
- LER 96-14-00	Loss of Process Monitoring Function During Postulated Fires (Appendix R)

Discussed/Updated

- URI 96-003-03	Smoke Detectors Added to CRVS Without 50.59 Evaluation

ATTACHMENT A

RESULTS OF A
DRCH/HOLB SIMULATOR EVALUATION
CONDUCTED AT INDIAN POINT 2

August 21-22, 1996

On August 21-22, 1996, NRC Operator Licensing staff from Headquarters and Region I visited Indian Point Unit 2 (IP2) to evaluate the performance and suitability of the simulation facility for Operator Licensing examinations and requalification training. The facility licensee was represented by Mr. Fehmi Aydin, Manager, Computer Applications/Simulator Program, and members of the simulator support staff including, computer applications management, operator training management, software engineers and testing personnel. The facility licensee has completed transition of IP2 training activities to the new simulator, which was certified in accordance with 10 CFR 55.45 by submittal of NRC Form 474, "Simulation Facility Certification," in November, 1995. The previous simulator is partially dismantled and is no longer in service. This limited evaluation concluded that the new IP2 simulation facility will fully support the requirements of operator licensing examinations and requalification.

BACKGROUND

A review of the IP2 simulator status and identified discrepancy reports (DRs) in June, 1996, raised concerns over the effectiveness of the simulator support program and suitability of the simulator for operator licensing examinations and licensed operator requalification training. These concerns included:

- The DR status report suggested that models are still being validated and in some instances are not fully integrated, processes that should have been completed by the simulator staff, not the trainees, before the simulator was certified to be operating in accordance with the Standard.

- The facility licensee staff appeared to have been unable to effectively schedule and resolve identified DRs in accordance with the procedures described in the simulation facility certification report.

- Priority 1 and Priority 2 DRs existed in virtually all areas of the simulation model. NSSS, BOP, and simulation-unique models and programs are equally represented in the update DR status report.

- The existing DR burden, coupled with the indicated sustained rate of new DR identification, potentially limited proposed examination scenarios to very narrow windows of operability. Even though scenarios might be pre-validated, the simulator, as described in this update status report, may not be relied upon to respond correctly to operator actions.

DISCUSSION

The objectives of the simulator evaluation visit were:

- to discuss and resolve NRC concerns.
- to observe simulator performance to confirm the ability of the NRC to use the IP2 simulator to conduct licensing examinations such that the requirements of 10 CFR 55.45 (b) may be met.

Entrance and exit meetings were conducted with the facility management. Meeting attendees are listed in the attachment to this report.

Simulator Evaluation Procedure

The NRC staff evaluated the Indian Point 2 simulation facility against the requirements of 10 CFR Part 55, using the criteria provided in ANSI/ANS 3.5, 1985 as endorsed by Regulatory Guide 1.149. The facility licensee certified the simulation facility to be maintained in accordance with these criteria using NRC Form 474, "Simulation Facility Certification," in November, 1995. The evaluation procedure comprised the following major areas, as described in NUREG-1258, "Evaluation Procedure for Simulation Facilities Certified Under 10 CFR 55."

- Performance Testing
- Design, Updating, Modification, and Testing

In accordance with the guidance of NUREG-1258, Section 3.1.1.3, Table 1, the NRC staff observed the following tests during the on-site review.

Normal Operations	Reactor startup - Test No. 14.3.7.2 Plant shutdown 100% to zero power - Test No. 14.3.7.5
Abnormal Operations	Dropped rod (stationary gripper) failure test - Test No. 14.3.8.8.2 Loss of shutdown cooling (RHR pump trip) - Test No. 14.3.9.22 Manual reactor trip - Test No. 14.3.9.23 Loss of service water - Test No. 14.3.9.26 Loss of CCW cooling - Test No. 14.3.9.27 Uncontrolled rod motion - Test No. 14.3.9.33 Inadvertent RCS dilution - Test No. 14.3.9.35
Emergency Operations	LOCA with blackout - Test No. 14.3.9.13 Faulted steam generator tube rupture exam scenario

As described in NUREG-1262, "Answers to Questions at Public Meetings Regarding Implementation of Title 10, Code of Federal Regulations, Part 55 on Operators' Licenses," no evaluation was made of the facility operators.

Concurrently, the NRC staff reviewed the disposition of the following fundamental system modeling and capability problems that had been identified during the DR status review: (The text in *italics* defines each DR as described by the simulator support staff in the DR tracking system.)

- *small break LOCA does not allow core reflood*

  Dry cladding to water heat transfer characteristics were modified to prevent excessive heat transfer. The post-LOCA reflood was retested.

- *the simulator blows up when going to recirculation*

  The event could not be reproduced by software engineers. The problem is thought to have been related to the heat transfer coefficient problem previously discussed.

- *reactor trip transient does not match plant response*

  The discrepancy was related to slow bypass valve response in the fast-opening mode, thus limiting the transient severity. Bypass valve stroke and pressurizer level response coding were modified and retested.

- *recirc pump flow lost when going to recirculation*

  Pump cavitation modeling was modified and retested.

- *service water flow needs to be tuned*

  Pump head/flow characteristics will be modified to match current plant data. Resolution is expected in the next software configuration to be released for training and examination (training load).

- *main turbine and MBFPs fail*

  Steam drain enthalpy calculations were modified and retested.

- *RHR flows unstable*

  This discrepancy was related to a previous discrepancy relating to cold-break LOCA. Reactor pressure vessel internal pressure calculations were modified and retested.

- *can't start 23 diesel*

  The problem was determined to be operator error.

- *core fails for no reason, gives no message*

  The problem was related to the heat transfer coefficient problem previously discussed.

- *break flow not interfaced to containment*

  Flow variable integration was established and retested.

The DR resolution evaluation included discussions with both the simulator software engineering and test operations personnel and a review of associated engineering and operational test documentation. In general the documentation of engineering resolution was adequate for continued simulator maintenance and configuration management. The staff noted that the documentation of operational retesting and revalidation was weak and would not easily support reconstructing a test environment. This discrepancy between engineering and operational documentation is considered by the NRC staff to be a weakness in that revalidation of software following significant modifications, as required by ANSI/ANS 3.5, cannot readily be confirmed by audit. The facility licensee staff was advised of this NRC concern.

The facility licensee explained that the initial NRC review of discrepancies had not taken into account two factors that were affecting management of the discrepancy resolution program. First, the simulator was still in a warranty status and the discrepancies were pending vendor warranty service when the initial NRC review was conducted. Second, many discrepancies remained in an open category pending incorporation in a training load, typically a six week cycle. A new training load was established shortly after the initial NRC review. Consequently, the burden of open discrepancies was noted to be substantially reduced at the time of this simulator evaluation.

The facility licensee further explained that automatic re-prioritization of discrepancies, the so-called "DR aging," had been found to be ineffective and was recently removed from the administrative procedures subsequent to the HOLB staff concerns. All discrepancies are now reviewed for priority and timeliness of resolution on a continuing basis by the simulator support staff.

The simulator provisions for examination security were reviewed. The instructor station computers are equipped with removable hard disc drives which can be secured to protect stored initial conditions or computer aided exercise scripts that may be generated during an examination preparation visit. Backtrack and replay files, unlike the instructor station scenario scripts, reside in the Encore main simulation computers which do not have removable storage media. The facility staff can erase backtrack and replay files as needed. No external modems or connections to other computer systems exist.

Each test was evaluated using the following criteria, excerpted from NUREG-1258. The numbers in parentheses relate to applicable sections of ANSI/ANS 3.5-1985. A determination was made of the impact of any nonconformance on the acceptability of the simulation facility for conduct of a licensing examination.

Parameter Relationships

- *Are expected relationships between this parameter and other parameters, according to the baseline data, reflected over the course of the performance test?*
  (3.1.1, 3.1.2 and A3.1)

The relationships between all parameters being tested were consistent with the baseline data to the extent that reference plant procedures could be followed. Plots of critical parameters were obtained for further comparison with reference plant and previous simulator performance data by the facility simulator support staff.

Alarms and Automatic Actions

- *Do all of the alarms and automatic actions occur that would have occurred in the reference plant?*
  (4.2.1(c))

  All alarms and automatic actions that were expected by the plant procedures occurred during the simulated events.

- *Do any alarms or automatic actions occur that would not have occurred in the reference plant?*
  (4.2.1(c))

  No unexpected alarms or automatic actions occurred during testing.

Transient Operations

- *If applicable reference plant start-up test procedure acceptance criteria exist, does the value represented by the parameter fall within these criteria?*
  (4.2.1(a))

  The procedures used were reference plant procedures as defined in the individual performance test procedure records. Critical procedural test parameters reflect current reference plant values. Simulator performance was consistent with procedural parameters.

- *Does the observable change in the parameter violate the physical laws of nature?*
  (4.2.1(b) and 4.2.2)

  The observed simulated plant response remained within the physical laws of nature.

- *Is the observable change in the parameter in the same direction as that expected from the baseline data?*
  (4.2.1(b) and 4.2.2)

  Observed changes in simulated parameters were consistent with baseline data and reference plant procedures.

Steady-State Operations

- *If it is a critical parameter, does it fall within ±2% of its reference value?*
  (4.1(3))

  All critical parameters that were monitored were within ± 2% of its reference value.

- *If it is a noncritical parameter, does it fall within ±10% of its reference value?*
  (4.1(3))

  One occurrence of RHR discharge pressure cycling was observed. The condition was documented and saved for further evaluation by the simulator support staff.

- *Has the accuracy of the computed values been determined for a minimum of three points over the power range?*
  (4.1)

  Initial simulator performance data includes validation of computed values over that full power range of operation. Performance plots are maintained on record in the simulator area and in the configuration management system.

- *For a 60 minute test, does the value of the parameter not vary more than $\pm 2\%$ over the 60 minute period?*
  (4.1(2) and A3.2(1))

  A 60 minute test was not within the limited scope of this evaluation and was not performed.

Simulator General Performance

The general performance of the simulation facility was evaluated using the following criteria. A determination was made of the impact of any nonconformance on the acceptability of the simulation facility for the conduct of a licensing examination.

- *Where applicable to the malfunctions tested, does the simulation facility provide the operator the capability of taking action to recover the plant, mitigate the consequences, or both?*
  (3.1.2)

  The facility staff operators were able to take all action specified in the appropriate reference plant procedures, including local operator actions performed from the instructor station.

- *For the performance tests conducted, is the simulation capable of continuing until such a time that a stable, controllable and safe condition is attained which can be continued to cold shutdown conditions, or until the simulation facility operating limits are reached?*
  (3.1.2)

  All tests were run to the completion of the appropriate reference plant procedures.

- *Does the simulation facility provide the appropriate response to operator errors, if any were tested?*
  (4.1(3), 4.1(4))

  No deliberate operator errors were tested.

- *Does the simulation facility respond inappropriately to any correct operator actions?*
  (4.1(3), 4.1(4))

  The simulator responded correctly to all operator actions.

- *Are there any differences identified between the procedures used in the simulation facility and controlled copies of reference plant procedures?*
  (A1.4)

  The procedures used in the simulator were the same as those used in the reference plant.

- *When tested by the staff, is simulation facility instrument error no greater than that of the comparable meter, transducer or related instrument system of the reference plant?*
  (4.1(1))

  The criteria for evaluating design, updating, modification, and testing, as given in this section, are based on the requirements of Sections 5 and A2(4) of ANSI/ANS 3.5.

Design Data

- *Do baseline data exist for all parameters tested?*
  (3.1.2, 5.1, A2 and A3.3)

  Baseline data for all tests are stored in the simulator area with the results of initial simulator acceptance testing.

- *If multiple sources of baseline data are available, are they used in the following order unless otherwise justified?*

  a. *Reference plant operational data - data collected directly from the reference plant.*

  b. *Analytical or design data - data generated through engineering analyses with a sound theoretical basis.*

  c. *Similar plant data - data collected from a plant which is similar in design and operation to the reference plant.*

  d. *Other data - data, such as subject matter expert estimates, which does not come from any of the above sources.*

  (3.1.2, 5.1, A2 and A3.3)

  The hierarchy of design and performance test data were not evaluated.

- *If the reference plant has been in commercial operation for 18 months, have plant data been included in the data base?*
  (5.1)

  Simulator design and performance test data includes reference plant operational data.

Updating and Modification

- *If 1) the reference plant has been in commercial operation for at least 18 months, and*

  2) it has been at least 18 months since the simulation facility's operational date,

  does the update design data base include actual plant data?
  (5.2)

  The simulation facility is within the first 18 months of operation.

- *Is there an annual review of reference plant modifications?*
  (5.2)

  Reference plant modifications since the initial construction design freeze have been reviewed and are being implemented in the simulator on a continuing basis.

- *Has the first such review been undertaken within one year of the simulation facility certification?*
  (Regulatory Guide 1.149, Section C, Item 4)

  Modification reviews have been completed within 1 year of simulator certification.

- *Have the simulation facility update design data been revised as appropriate, based on an engineering, training value, and licensing examination assessment of the reference plant modifications identified in the annual review described in item 2 above?*
  (5.2)

  The simulation design data base is updated on a continuing basis using the configuration management system.

- *Is there a means of incorporating student feedback on the simulation facility into the updating and modification process?*
  (5.2)

  Students are encouraged to provide feedback using the discrepancy reporting system.

- *Have all modifications to the simulation facility required as a result of the assessment performed in Item 3 above, been made within 12 months of their identification?*
  (5.3)

  All required modifications to the simulation facility have been incorporated or are currently scheduled for incorporation in future training loads.

Testing

- *Are data from simulation facility performance tests which were performed after completion of initial construction and after any configuration or performance modifications available for review?*
  (5.4.1 and A2(4))

  Design data and performance test results data are maintained in hard copy in the simulator area. The configuration management system is also accessible from the instructors station.

- *Are data from the annual operability testing available for review?*
  (5.4.2 and A2(4))

  Initial operability testing data is maintained in the simulator area. The annual operability testing has not yet been required.

Known Discrepancies

Discrepancies identified during the course of this simulator evaluation which were previously known to the facility licensee and for which resolutions or justifications were provided were reviewed.

- *Could any of the discrepancies have a significant adverse affect on the conduct of a licensing examination?*

  No priority 1 discrepancies existed at the time of the simulator evaluation. Nine priority 2 discrepancies existed and they were all expected to be resolved with the next training load. The priority 2 and priority 3 discrepancies were reviewed and found to have little or no potential adverse effects on the conduct of operator licensing examinations.

- *Are there any facility licensee resolutions or responses with which the staff does not agree?*

  The NRC staff has no disagreement with the facility licensee's discrepancy resolution or simulator support activities.

INDIAN POINT 2 SIMULATOR EVALUATION
MEETING ATTENDANCE

August 21-22, 1996

Entrance Meeting

NRC
Frank Collins
Brian Hughes
Carl Sisco

Consolidated Edison
John Ferrick
Vic Mullen
Kuo-Chun Chi
Fehmi Aydin
Steve Quinn
Charles Jackson
Richard Louie
John Ellwanger
J.C. Rowland
Gary Keene

Exit Meeting

NRC
Frank Collins
Brian Hughes
Carl Sisco
Robert Temps (Senior Resident Inspector)

Consolidated Edison
John Ferrick
John Weiss
Kuo-Chun Chi
Fehmi Aydin
Steve Quinn
Charles Jackson
Richard Louie
John Ellwanger
J.C. Rowland
Gary Keene
Bill Kriebel

ENCLOSURE 2

Management Meeting

Indian Point 2
Docket 50-247

September 12, 1996

Agenda

1. *Introduction & Overview*

2. *Valve 863 Design Description & Licensing Basis History*

3. *Event Timeline*

4. *Operator Actions*

5. *Station Nuclear Safety Committee Review*

6. *Further Corrective Actions*

7. *Summary & Closing Comments*

Valve 863 Design Description

One Inch Diameter Globe Valve

Air Diaphragm Operator Supplied With Instrument Air (Fails Closed)

Remote Manual Operation From The Central Control Room

Located On Elevation 80 Ft In The Primary Auxiliary Building (Accessible Post Accident)

Containment Isolation Valve In The High Pressure Nitrogen Line To Containment That Serves

- *The Four ECCS Accumulators (Primary Supply)*
- *The Power Operated Relief Valves On The Pressurizer (Primary Supply)*
- *The Pneumatic Instrumentation For The Alternate Safe Shutdown System(Fire Protection) (Backup Supply To Instrument Air)*

Licensing Basis History

1970	*Original FSAR: Valve 863 Shown Normally Open* (Table 5.2-1)
1976	*NRC Initiates Review Of Proposed Technical Specification Changes For 10CFR50 Appendix J: Valve 863 Listed As Open Continuously Or Intermittently For Plant Operation (Tech Spec Table 3.6-1)*
1977	*PORV/LTOP Design Change Description Submittal Based On Nitrogen Bottles As Normal Supply (Valve 863 Has To Be Open) And Added Check Valve 4312*
1980	*NRC Review Of TMI Submittal: Valve 863 Shown Normally Closed But Could Be Used Post Accident* *Technical Specifications Issued For Appendix J With Valve 863 On Table 3.6-1*
1981	*Technical Specification 3.1 Requires PORVs Operable When RCS Is Above 350°F*
1982	*First FSAR Update: Valve 863 Shown Normally Closed In Section 5.2 (Based on TMI Submittal). Figure 6.2-1 Shows Normally Open And Section 4.3 Describes PORVs Normal Supply Of Nitrogen As Bottles Banks Outside Of Containment (Valve 863 Has To Be Open)*

1984 *NRC SER For LTOP: Describes PORVs Primary Supply Of Nitrogen As Bottle Banks Outside Of Containment (Valve 863 Has To Be Open)*

1985 *Technical Specification 3.16 Requires PORVs Operable When RCS is Above 350°F*

Technical Specification 3.1 Issued To Incorporate LTOP Requirements

Event Timeline

July 3, 1996	*System Engineer Identifies Document Discrepancy Regarding Containment Isolation Valves - Open Item Report Initiated*
July 5, 1996	*Three Additional Open Item Reports Initiated Regarding Containment Isolation Valves In Response To Questions From The Resident Inspector*
July 8, 1996	*Generic Review Of UFSAR Table Initiated*
July 11, 1996	*NRC Inspector Questions UFSAR Discrepancy For Valve 863*
	Valve 863 Closed Per SOP Until Documentation Discrepancy Is Resolved
	Nitrogen Header Low Pressure Alarms Received In The Central Control Room
	Documentation Reviews Conducted - FSAR, Technical Specifications, DBD, System Description And Safety Evaluations For LTOP Modification
	Open Item Report Initiated For Valve 863 Discrepancy
	Station Nuclear Safety Committee Reviews And Approves Procedure Changes To Allow Valve 863 To Remain Open

July 12, 1996	*Con Edison Management Recognizes That A Safety Evaluation For The Emergency Procedure Change Was Not Performed As Required By Administrative Procedures*
	Containment Entries Performed To Identify And Correct Several Leaks
	Conference Calls With NRC On Valve 863
July 12 - 18	*Open Item Reports Initiated For Various Containment Isolation Valves To Document Discrepancies Found Between UFSAR, Tech Specs, IST Program, And/Or Station Procedures*
July 13	*Detailed Licensing Basis Re-review*
July 15	*Con Edison Task Force Formed To Expeditiously Resolve Discrepancies*
July 18	*Operability Reviews Completed And On File*

Operator Actions

Operators Believed Valve 863 Should Be Open

Reinforced Management's Expectations And Standards

Conducted Extensive Review Of Operational Tasks Considered To Be Routine Evolutions

Conducted Quality Assurance Surveillance Of Safety Significant Checkoff Lists

Created Task Review Sheet To Evaluate Operational Activities Performed On-Shift vs. Procedural Guidance

Revised Policy For Operations Procedure Adherence

Conducted Evaluation Of Mechanisms Available To Generate Procedure Changes And Enhancements

Station Nuclear Safety Committee Review

July 11, 1996 Meeting:

- o *Technical Specification 3.6 Permits Valve 863 To Be Open Continuously Or Intermittently For Plant Operation*
- o *Detailed Knowledge Of Licensing Basis As A Result Of Research And Discussions Preceding The Meeting*
- o *Reviewed The TPCs, Prior Safety Evaluations For LTOP, The UFSAR*
- o *Institutional Knowledge Of The Tech Spec History*
- o *Open Position For Valve 863 Was Believed To Be Consistent With The Current Licensing Basis (LTOP)*
- o *Since Tech Specs Addressed And Permitted Action, An Unreviewed Safety Question Did Not Exist*

Further Corrective Actions

Documented Results Of Task Force Review Of Containment Isolation Valves

Implemented Changes To Facility Documents And Prepared Written Safety Evaluations Where Required

Conducting A Review Of UFSAR To Identify Other Discrepancies
- *Resolve With Appropriate Action Including Operability Determinations, Written Safety Evaluations, And Changes To Facility Documentation*

Summary & Closing Comments

Operators Believed That It Was Acceptable For Valve 863 To Be Normally Open

Personnel Error Resulted In The Failure To Have A Safety Evaluation For An EOP Change, As Required By Plant Administrative Procedure

Procedural Adherence Issue Resulted From Not Having The Procedure In Hand

Safety Committee Review Was Based Upon The Tech Specs Permitting The Operation Of Valve 863 In The Open Position And Thus Did Not Question Further The Need For A Written Safety Evaluation

Three Parallel Issue Submittal And Review Paths In The Late 1970s And Early 1980s Affected Valve 863 Designated Position

- *The Third And Last Path Was Based Upon The Valve Being Normally Open To Supply Nitrogen For PORV Operation*

- *Two Of These Paths Called For Or Were Based Upon An Assumed Normally Closed Position*

A Comprehensive Review Of <u>ALL</u> Containment Isolation Valves Was Conducted

An Engineering Review Of UFSAR Is Being Performed